X-17426

UNCLASSIFIED

19970311 143

AECD-3712

Subject Category: PHYSICS

UNITED STATES ATOMIC ENERGY COMMISSION

HISTORY AND STATUS OF THE EBR

By
Warner E. Unbehaun

April 15, 1953

Chicago Operations Office
Lemont, Illinois

ATOMIC ENERGY COMMISSION
UNITED STATES OF AMERICA

Technical Information Extension, Oak Ridge, Tennessee

UNCLASSIFIED

Date Declassified: November 14, 1955.

Printed in USA, Price 40 cents. Available from the Office of Technical Services, Department of Commerce, Washington 25, D. C.

AECD-3712

HISTORY AND STATUS OF THE EBR

by

WARNER E. UNBEHAUN

Reactor Division
Chicago Operations Office
U. S. Atomic Energy Commission

April 15, 1953

T A B L E O F C O N T E N T S

HISTORY AND STATUS OF THE EBR

Introduction

The feasibility of constructing a nuclear reactor operating in the fast neutron spectrum for the production of fissionable material and power has been under study at Chicago since 1945. It is hoped that such reactors can eventually be constructed to economically convert fertile material into fissionable material at a rate significantly greater than it will be consumed and simultaneously produce significant amounts of electrical power. The Argonne National Laboratory has built and has been operating since August, 1951, a 1,000 kw fast reactor known as the Experimental Breeder Reactor (EBR). The machine was primarily designed to perform limited experiments using the smallest possible critical mass.

Work on various types of fast breeders has been studied at KAPL, and Los Alamos is now operating a mercury cooled Pu fueled fast reactor. Brookhaven has been studying a lead cooled unit. This report will attempt review of the program and progress which has been made to date on the EBR.

History

The first concept of a fast reactor was considered by Dr. W. H. Zinn and Dr. Enrico Fermi when they discussed the feasibility of a fast reactor consisting of a solid mass of fissionable material, penetrated with coolant channels to remove the heat produced during the chain reaction process.

Work on the Experimental Breeder Reactor was initiated early in 1945. The preliminary basic work consisted of metallurgical studies, investigation of the properties of liquid metals, heat transfer, fast cross section studies, and basic nuclear studies. Zinn and Mr. Harold Lichtenberger worked on preliminary design studies; Meyer Novick on heat transfer; Frank Foote on corrosion and metallurgy; Arthur Barnes on liquid metal pump development; and Leonard Koch on engineering; Frank Hoyt, Maria Mayer, Elmer Eisner, Gale Young, Robert Sachs, Mazopski, Jane and David Hall were active at various times in performing the theoretical work. Louis Slatin performed the critical mass experiments for the EBR at Los Alamos in 1946.

The liquid metals work was done at Site B in the city of Chicago, and at Site A at Palos Park in a specially constructed facility known as Building K. The cost of Building K was $26,000. It was fortunate that a commercial supplier for the NaK alloy was available. It was found that the Mine Safety Appliance Company was

using the alloy as a reagent to produce oxygen in fire fighting gas masks. This company was familiar with the technology of producing the metals and supplied all that was required by Argonne. Philip Abelson of Mine Safety Appliance Company played an important role in supplying ANL with the required NaK.

By the end of 1945, preliminary sketches for the design had been prepared by Zinn and Lichtenberger and it was proposed that an experimental test pile be constructed. The geometry of the reactor consisted of closely packed cylindrical fuel rods with the coolant flowing through the space between the rods. The coolant, sodium-potassium alloy, was favored and it was decided that the use of U^{235} initially and Pu later would be employed as the fissionable materials. The breeding gain was roughly estimated on the basis of data available to be about 0.2.

The Commission approved the design and construction of the reactor on November 19, 1947. On January 25, 1948, the Reactor Safeguard Committee advised that a power level of 1,000 kw/hr average and of Pu content of not more than 250 grams for the EBR would not create an undue hazard if located at the DuPage Site.

On April 7, 1948, approval was given by the Commission to construct the EBR and its associated chemical and metallurgical

processing plants at the DuPage Site of the Argonne National Laboratory, subject to the conditions set forth by the Reactor Safeguard Committee. It was recognized, however, that the location of the reactor at a remote testing station was a possibility. The selection of the Austin Company of Cleveland, Ohio, as the architect-engineer for the detailed design of the building and some of the equipment was approved on January 25, 1948.

In a letter dated February 26, 1949 to Dr. Hafstad, Dr. Zinn requested that the location of the reactor be changed to the National Reactor Testing Station in Idaho to permit greater flexibility in operation. The reason for this request was that it appeared both desirable and feasible to operate at a power level greater than 1,000 kw and with a greater Pu content than 250 grams. In July, 1949, the General Advisory Committee concurred in the construction of the EBR at the National Reactor Testing Station and in September, 1949, the Commission formally approved this action.

In November, 1949, contract negotiations for construction of the reactor building and its supporting facilities were initiated with the Bechtel Corporation of San Francisco.

Design and construction proceeded concurrently and in April, 1951, the reactor, auxiliary equipment, and buildings were complete.

The reactor was loaded and found to be sub-critical with the amount of fuel originally provided. Consequently, more fuel was added to the core and on August 24, 1951, the machine went critical with 51.8 kg of U^{235}. On December 20, 1951, the reactor was used as a heat source to produce a significant amount of electricity from nuclear energy for the first time in history. The following day, the supply of conventional electricity to the building was cut off and the entire power load was carried by the electricity supplied by the nuclear source and its associated equipment when 150 kw of electrical power were generated. Figure 1 is a photograph of an inside wall of the reactor building showing the names of those present when electricity was first produced.

The EBR was primarily designed as an experiment to test the possibility of breeding, to gain experimental data in handling radioactive liquid metals at high temperatures, and to obtain data peculiar to the operation of nuclear reactors operating in the high energy neutron range. The power production is incidental in the operation of the reactor. It is, therefore, not intended that a comparison should be made of the cost of producing electrical power in the EBR with conventional methods. However, technical information will be gained that may be useful

in developing high power reactors capable of producing electrical energy at a cost competitive with conventional sources.

The EBR was the first budgeted reactor; that is, it was the first reactor for which the Atomic Energy Commission budgeted a fixed amount of money for the complete facility including the building for the reactor and auxiliary facilities. The reactor and reactor vessel, the reactor control system and instrumentation, and special coolant system components were constructed by Argonne National Laboratory. The latter includes electro-magnetic and mechanical pumps, flowmeters, pressure gauges, liquid level indicators, and the steam generator. The building and auxiliaries were constructed by the Bechtel Corporation of San Francisco. Total construction cost is estimated to be about $2,700,000. It is estimated that approximately $2,500,000 was spent by the Laboratory over a four-year period for research and engineering development.

Theory and Status of EBR

The fast reactor is designed to be self-sustaining on neutrons in the high energy range. Advantages of operating a reactor in the high neutron energy range will be discussed in the following paragraphs.

When an atom fissions, a certain number of neutrons are emitted. The number of neutrons emitted per fission, ν, is a statistical value and is not a true representation of the number of usable neutrons per neutron absorbed for maintaining the chain reaction and production of additional fissile material. Some neutrons are absorbed in the fissile material without causing fission and result in the loss of the fissile atom. Thus, for example, Pu^{239}, is transmuted to Pu^{240}. The ratio of the number of neutrons that are captured parasitically in the fissile material to the number that are absorbed to cause fission is designated as the α of the material. In the intermediate energy range, the values of α are not well known due to resonance peaks. It appears, however, that α increases with neutron energy to a certain point and then reverses with further energy increases, becoming smaller at the fission energy spectrum than in the thermal range. Therefore, operating in the intermediate energy spectrum does not appear attractive for breeding, whereas operation in the high energy range for all fissionable materials

and in the thermal range for U^{233} does offer possibilities.

This, then introduces the concept of a new value, η which takes into account α . η is defined as the number of neutrons produced per neutron absorbed in fissionable materials. Some approximate values of η for various fuels and for specific neutron energies are listed below:

Neutron Energy	U^{235}	U^{233}	Pu^{239}
Thermal	2.10	2.36	2.05
Fast	2.5	2.65	2.95

Some of the values listed above are probably somewhat higher than actual, since recent experiments indicate that α is much higher than originally expected.

Another parameter which is most favorable in a fast reactor is the contribution of fissions in U^{238} caused by neutron reactions. Normally, when U^{238} absorbs a neutron it is transmuted to Pu. However, at high energy levels above a fixed threshold value, some neutrons cause fission when absorbed by a U^{238} atom. The fast effect, ϵ , may be defined as the ratio of the neutrons produced due to fission of U^{238} and U^{235} to the neutrons produced due to fission in U^{235} only. ϵ is thought to be about 1.05 maximum in the thermal range and as high as 1.15 maximum in the fast range. η eff., or $\eta\epsilon$, could be considered as a factor that gives a more precise but, of course, not perfect picture of the net number of neutrons available per neutron absorbed in fuel neglecting contributions from other fissile isotopes such as Pu^{241} which will be created in small quantities by absorption processes.

The potential values of $\eta_{eff.}$, therefore, could be as high as:

	η eff.		
Neutron Energy	U^{235}	U^{233}	Pu^{239}
Thermal	2.2	2.48	2.15
Fast	2.88	3.05	3.4

It is, therefore, apparent that the neutron energy spectrum in a fast reactor should be depressed as little as possible in order to obtain the greatest yield. Therefore, moderator is not used and all structural materials and coolants are selected which have small scattering as well as low absorption properties. However, at high energies the cross section for the fuel is also low. It, therefore, requires large fuel inventories for a given fission rate. Thus fast reactors are characterized by lower specific power and higher critical masses when compared with thermal reactors. A fast reactor may be operated either as a breeder or converter. A true breeder is a reactor which produces a greater quantity of the same type of fuel that it burns up by the process of transmuting fertile material. For example, in a Pu^{239} breeder, the fuel burned is Pu^{239} and the fertile material is U^{238} which is transmuted to the Pu^{239}. A similar cycle could be used for U^{233}. A converter burns one fissile fuel to produce another product which may be either another fissile fuel or some other desirable product. For example, a

converter would burn U^{235} to either produce Pu^{239} by absorption of neutrons in U^{238}, or, if required, it may produce U^{233} by absorption of neutrons in Th^{232}, or tritium by absorption of neutrons in lithium.

The breeding gain is defined as the number of excess fissionable atoms produced per atom burned. In simplified form, it is determined by the relation

$$B.\ G. = \eta_{eff.} - 2 - \text{losses}$$

where B. G. is the breeding gain, $\eta_{eff.}$ is the number of neutrons produced per neutron absorbed in fuel including neutrons obtained from fast fissions in U^{238}, and the losses refer to neutrons absorbed in poisons, structural material, etc., and leakage from the reactor. The factor of 2 comes from the fact that one neutron per fission must be absorbed in another fuel atom to keep the chain reaction going and one must be absorbed in the fertile material to produce the same amount of fuel as burned. The breeding ratio, (B.R.) is the total number of fissionable atoms produced per atom burned and is determined by the relation B.R. = $\eta_{eff.}$-1-losses. In a converter, this value is termed the conversion ratio.

Another parameter indicating the merits of a breeder is the doubling time, i.e., the time required to double the original

fuel investment. This is a measure of profit. Doubling time, neglecting "out of reactor" inventory, is defined by the relation

$$\text{Doubling Time (Days)} = \frac{10^6}{\text{Breeding Gain x Specific Power}}$$

The specific power may be increased in a fast reactor by diluting the fuel with U^{238}. Diluting serves two purposes. First, it increases the surface to volume ratio of the heat producing volume so that a higher specific power can be achieved. Second, it places U^{238} or fertile material in the reactor core where it can most effectively compete for the capture of fast neutrons to cause fission or be absorbed and transmuted to plutonium and thus add to the breeding gain. The amount of diluent that is permitted in a fast reactor, however, is limited since an excess amount will depress the neutron spectrum by elastic and inelastic scattering and increase the critical mass by expanding the core volume. There is some question as to the most optimum ratio of fertile material to fissionable material in the core.

Another factor affecting the efficiency of a fast breeder is the cost and potential losses in chemical processing. These losses can be held to a minimum by achieving as high a burnup as possible and, therefore, decreasing the frequency of the chemical processing. Burnup of 10% of the enriched fuel is believed the minimum for economical operations.

An economical fast breeder requires the use of materials having the high heat transfer properties of a liquid metal coolant and non-moderating properties, hence NaK was chosen as the coolant for the EBR. Pure Na would be superior but it is inconvenient due to its higher melting temperature and the need for auxiliary heating.

The fast reactor cycle is not limited to any one fuel since values of η are high for all fissionable materials. Plutonium appears to have the highest worth of the three fissionable materials. Thus, as a converter, the reactor would burn the least valuable fissionable material and produce the most valuable fissionable material, as in a U^{235}-Pu^{239} converter. Breeding, on the other hand, provides a method for increasing by a factor of 30-40 the yield of fissionable material or up to 140 times the heat energy from natural uranium.

Summarizing, the advantages of a fast reactor are:

1. <u>Greater Neutron Yield</u>

   Higher net yield of neutrons for all fissionable materials, hence, a greater net availability of fissionable materials and power from a limited ore supply is possible.

2. <u>Greater Selection of Materials</u>

   At fast energies, absorption cross sections for most structural and coolant materials are all relatively low so a greater selection is available. However, there are limitations insofar as moderation and scattering of neutrons are concerned.

The disadvantages of a fast reactor are:

1. <u>High Inventories of Fissionable Materials</u>

   The specific power appears to be limited to lower values than is possible with thermal reactors and hence high inventories are required. In addition, at high neutron energies, the cross sections for fissionable materials are lower than at thermal energies. Thus a larger critical mass and inventory of fissionable material are required.

2. <u>Liquid Metal Coolant Required</u>

   Due to fuel dilution limitations, and nuclear requirements, a liquid metal coolant is necessary to produce the high power densities required. Liquid metals usually involve extra safety hazards.

3. <u>Cost</u>

   There is less experience and technology available, hence, progress and costs can be expected to be less favorable for immediate applications. Costs, in the future for fast reactors, may be comparable with other reactor types, or perhaps even more favorable.

The place of the power breeder reactor is at present dependent primarily upon economics since there are other methods for achieving production of the desired fissionable materials and electricity. Present design data indicate that it will require approximately 300-500 kgs of U^{235} or 200-300 kgs of Pu to charge a 500 to

1,000 MW machine. Holdup of spent fuel in inventory, of course, will increase the total fissionable material requirements depending upon attainable burnups. The economic desirability for a power breeder, therefore, tends to pivot on the balancing of costs and availability of uranium ores, demand and the value of plutonium, and the efficiency of the thermal cycle, since with liquid metal coolants, it is possible to increase the electrical output for a given heat output.

If U^{238} thermal reactors are used for straight power production, i.e., high burnup in the fuel, the Pu thus produced may be efficiently used as a cheap fuel supply for power breeders.
The work that can be done with the EBR is limited due to its size. The core is too small to effectively permit the study of dilution factors. Due to the undertainties of the present values which may be peculiar to the EBR geometry, it would not be advisable to consider extrapolating the present nuclear values for the design of a large machine.
The value of the conversion ratio in the EBR is now being determined to give an indication of the economics and production capabilities of such a machine. This is being done by essentially two methods: chemical analysis and by nuclear measurements of neutron distributions. Nuclear measurements can

provide preliminary indications of breeding, but actual chemical separation of the materials will determine the value more accurately. This will require a considerable operating period for the reactor.

There is insufficient data on the irradiation exposure of materials, particularly fuels in a fast neutron flux, and considerable effort will be required to investigate irradiation effects in the high energy level in the EBR. Long term irradiations of the coolant should also be made in order to determine if there are appreciable long-lived activities generated.

Description of the EBR

The EBR is presently charged with enriched uranium-235 as fuel and is cooled with NaK alloy. It operates in the fast neutron energy range and, hence, contains no moderator.

Figure 2 is a photograph of the outside of the building and Figure 3 shows a cross section of the reactor. The active core originally consisted of .364" diameter U^{235} rods closely spaced in a hexagonal pattern $7\frac{1}{2}$" across the flats and $7\frac{1}{2}$" high and was designed to hold 40 kg of fuel with the outer row of rods filled with natural uranium. This outer row provided space for eight additional kilograms of fuel in the event it would be needed to achieve criticality. It was found later that 51.8 kg of fuel were required for criticality.

This not only made necessary the replacement of the natural uranium in the outer row of U^{235}, but also made it necessary to use some rods of .384" diameter and somewhat longer. This change aided in flattening the flux and changed the ratio of maximum to average flux in the core from 1.50 to 1.25. The inner blanket surrounds the core and is made up of rods of natural uranium .894" in diameter also closely packed. There is also natural uranium above and below the core. The diameter of the inner blanket is 15-7/8" and it extends above and below the core 8" and 4-3/4" respectively. Details of the fuel and fertile rods are given in the section on metallurgy.

The core and inner blanket are contained in a stainless steel tank with a top and bottom plate in which the rods of fuel and fertile material are engaged. NaK coolant flows between the tubes in the core and inner blanket. Figure 4 is a photograph of the reactor inner assembly. Figure 5 shows the top plate, and Figure 6 the bottom plate with the rod engaging holes. Figure 7 shows the bottom plate with tie rods which bolt to the shield section of the assembly and the hexagonal flow separator that separates the core and inner blanket. The NaK coolant enters the reactor at the top, flows down around the internal blanket rods, through the bottom plate and up around the rods in the active core.

Outside of the inner tank is the external blanket. The external blanket consists of natural uranium bricks clad in stainless steel. Twelve bricks per layer are stacked seven layers high around the periphery of the inner tank, making an external blanket 30-7/8" in diameter and about 27" high. The bricks are cooled by air which flows through five finned holes in each brick. Each brick also contains one hole providing a space for a total of eight external safety rods and four external control rods. Figure 8 is a photograph showing the top and bottom of a blanket brick and Figure 9 shows the bricks assembled in place around the inner liner to make up the external breeding blanket. The natural uranium bricks that make up the external blanket are assembled on an elevator platform that can be raised and lowered, and is used as a shim control.

Centered at the base of the external blanket and directly beneath the reactor tank is a safety plug. This is a movable block of natural uranium weighing 81 kg and jacketed with stainless steel and held in position by air pressure. In the event of a power failure, the valves assume the open position and drive the plug down away from the reactor.

Surrounding the external blanket is a graphite reflector 19" thick and finally about 9' of concrete shielding. Six experimental beam holes pierce the concrete shielding and graphite

reflector to provide facilities for instrumentation and irradiation. Also provided is a thermal column, and a rabbit hole that goes through the shield tangent to the exterior of the outer blanket. Figure 10 is a photograph looking vertically downward at the reactor shield and top of the reactor tank. The large air duct in the background is for cooling air for the external blanket.

Unloading of the spent fuel elements is achieved by placing a large torpedo-shaped lead coffin over the top of the reactor. By proper adjustment of the coffin and an off-center opening in the top plate any fuel tube may be selected and withdrawn into the coffin. After the tubes are withdrawn they are transferred to a shielded wash cell where the NaK adhering to the external surfaces of the tube is removed. The rods are then stored to permit "cooling" of the short-lived radioactivity after which they are transferred to a shielded cell where the stainless steel tube is cut both longitudinally and around the periphery of the slugs and the slugs removed. Figure 11 shows an operator adjusting the mechanism which carries the ten ton removal coffin over the reactor in preparation for removing an irradiated fuel rod.

The composition of the NaK coolant is 22.5% sodium and 77.5% potassium by weight. It has a low melting point (-12.3°C) and a high boiling point (783°C.) The specific heat is .212 cal/gm/°C.

Although sodium has a much higher specific heat (.3055 cal/gm/^{o}C at 400^{o}C) which favors higher heat transfer rates, its high melting point (98^{o}C) is a disadvantage because of the freeze-up problems. Corrosion rates at elevated temperatures are tolerable for a number of common structural materials including nickel iron and 347 stainless steel. Sodium-potassium does not possess significant moderating properties, and the NaK system operates at low pressure. All of these factors contribute in making NaK an attractive coolant, particularly for fast reactors. Its principal disadvantage is its violent reaction when brought in contact with water. Hence, the need for an inert gas atmosphere over all tanks, the top of the reactor, pumps, and in the irradiated fuel handling gear.

The nominal design power of the reactor is 1000 kw, however, many of the components of the system are capable of operating at some increased value. The operation of the reactor is presently limited to 1400 kilowatts by the heat removal capacity of the flowing air in the external blanket.

Figure 12 is a flow sheet of the primary and secondary coolant system. The NaK in the primary system is contained in a 3000 gallon gravity feed tank. It flows from this tank by gravity through the reactor, into an expansion tank, through the primary heat exchanger, and finally into a 1000 gallon receiver

tank. It is then pumped from the receiver tank back up into the storage tank by a 500 gpm electromagnetic pump at a 60' head. A mechanical pump of comparable capacity is connected in the flow system in parallel with the electromagnetic pump, and is normally idle, being maintained as a standby unit. A 4000 gallon drain tank is provided for removing the NaK from the primary system. The NaK becomes radioactive when it flows through the reactor, consequently, the primary circuit is shielded.

The heat in the NaK is transferred to a secondary NaK system in the primary heat exchanger. The NaK in the secondary system is pumped into a steam generator by a 360 gpm 60' head centrifugal pump where 550°F and 400 psig superheated steam is generated. The steam generator consists of three sections--the economizer, the evaporator, and the superheater. Heat exchange is to a falling water film in the evaporator, to an annular baffled space containing water in the economizer, and to steam in the superheater. Tubes in the three steam generator components consist of two concentric nickel tubes with a copper annulus between them which has been diffusion bonded to the nickel surfaces to insure good heat transfer. Figure 13 is a photograph of the steam generator and Figure 14 is a schematic drawing showing the coolant flow and temperatures at various locations in the system. Figure 15 is an artist's sketch of a

section through the building showing the equipment layout. Figure 16 is a picture of the control room.

For low power operation, the heat is discharged by an air cooler. The following is a summary of the nominal design specifications for the reactor:

1. GENERAL

Heat power output	1000 kw
Electrical power output	170 kw
Generator capacity	250 kw
Specific power	20 kw/kg
Power density in core	250 kw/liter
Coolant	NaK-22.5% Na, 77.5% K by weight 50% Na, 50% K by volume
Coolant flow rate	210 gpm
Fuel	about 90% enriched uranium
Steam temperature	550°F
Superheat	100°F
Steam pressure	400 psig

2. CORE

Approximate core size	Hexagon $7\frac{1}{2}$" across flats x $7\frac{1}{2}$" high
Critical mass (cold, clean)	51.800 kg
Provision for excess K	3.875 kg
Total mass in core	55.675 kg

Slug diameter	Some .364"; some .384"
Liquid bond thickness	Some .020"; some .010"
Outer S.S. tube wall thickness	.022"
Outside diameter of outer tube	.448"
Mass of U^{235} in each tube	223 grams in small size
	333 grams in large size

3. INTERNAL BLANKET

Material	Natural uranium rods
Outside Diameter of blanket	15-7/8"
Height	20¼"
Slug diameter	.894"
Outer S.S. tube wall thickness	.022"
Outer diameter of outer tube	.938"
Weight of uranium	1187 lb.

4. OUTER BLANKET

Material	Natural uranium bricks
Outside diameter of blanket	30-7/8"
Thickness of blanket	6½"
Weight of uranium	8775 lb.
Jacket thickness	.020"

5. PHYSICS DATA

Conversion ratio (preliminary measurements)	1.0
Average neutron energy	1.0 Mev
Neutron flux:	
At center of core	0.936×10^{14} n/cm^2/sec
At edge of core	4.86×10^{13} "
Between internal and external Blanket	1.9×10^{12} "
At outer edge of outer blanket	1.2×10^{11} "
Ratio of maximum to average flux in core	1.25

6. COOLING CIRCUIT (See Figure 14)

a. Primary System:	
NaK inlet to reactor	250°C
NaK outlet from reactor	350°C
Flow rate	210 gpm
b. Secondary System:	
NaK inlet to primary heat exchanger	241°C
NaK outlet from primary heat exchanger	340°C
Flow rate	210 gpm

c. Steam generator (economizer, evaporator, and superheater)

NaK inlet temperature	340°C
NaK outlet temperature	242°C
Flow rate	210 gpm
Water inlet temperature	220°F
Steam outlet temperature	550°F

d. Capacity:

Economizer	1,432,000 Btu/hr.
Evaporator	4,560,000 Btu/hr.
Superheater	368,000 Btu/hr.

7. <u>REFLECTOR</u>

Inside diameter	38-7/8"
Thickness	19"

Component Design

During the course of the design of the EBR many unique engineering features peculiar to the requirements of the EBR were developed at Argonne. Such developments are in the fields of metallurgical processing and fabrication techniques for fuel elements, liquid metal techniques, instrumentations, heat transfer and heat exchangers, valves, pumps, remote handling techniques, and many others.

The pumps used in the EBR coolant system are a good example of some of the many developments achieved. A totally enclosed liquid metal mechanical pump was developed. This pump is divided into two sections. The upper section contains the motor, bearings and bearing support while the impeller is supported in the lower section by a long shaft projecting down from the upper section. The pump is designed in such a manner that the height to which the liquid can rise in the tank is limited so that it can never contact the seal. Hence, only the argon gas blanket need be sealed and there is no chance of a sodium leak.

The sodium is brought out through the lower tank by means of a bellows seal. A fan circulates air through an annular space around the motor compartment to cool the pump. Capacity of this pump is 360 gpm and it operates with a 60' head. Two of these pumps are used in the system.

A special electromagnetic pump was also developed for the EBR at Argonne. It is a D. C. linear conduction type with the magnetic field winding connected in series with the current path through the liquid. The entire pump tube and magnet assembly is completely enclosed in a welded stainless steel box which is pressurized with argon. Stainless steel bellows serve as insulating seals through which the heavy copper leads are brought into the enclosure.

Current is supplied by a water cooled copper sulphide magnesium rectifier that has a maximum capacity of 20,000 amps at 1.0 volt and at this point operates at an efficiency of 40%. Control is by means of a separate autotransformer. The capacity of this pump is in the range of 400-500 gpm at 60' head.

It was necessary to design special valves that insured against any possible leakage of the NaK coolant into the atmosphere. The final design arrived at contains a double bellows seal, one contained within the other. The bellows and the body of the valve are made of 347 stainless steel and are entirely welded. A probe is connected to an electrical circuit to signal if the sodium should leak from the system through the inner bellows and into the chamber between the bellows. If the outer bellows developes a leak the pressure between the bellows would auto-

matically increase and such action would be detected by instruments provided. Therefore, only failures in both bellows occurring at the same time could result in a leak of the NaK to the outside. This unique feature has proven extremely successful and has been duplicated by others in designing reactors where leakage from the system is a serious problem.

A unique method of preventing a leak of the NaK coolant into the water or steam in the steam generator was developed. The tubes in the steam generator are made up of concentric assemblies of two nickel tubes with a copper annulus between them. The copper annulus contains small grooves through which a gas is passed. After passing through the tubes the gas is introduced into a blue flame. If a leak develops in the tube, sodium vapor from the coolant will be picked up by the gas system and change the flame to a brilliant yellow indicating the leakage of NaK through the outer nickel wall. At that point the alloy still has the copper wall and inner nickel wall to penetrate before reaching the water.

In the NaK to NaK heat exchanger, the pressure in the secondary non-radioactive side is kept higher than that in the primary radioactive side. A leak in the system is indicated by a diminution of material in the secondary circuit.

It was necessary to design unconventional pressure gauges for the coolant system that would not be affected by temperature and that would operate in areas of intense radiation. A pressure gauge was developed that utilizes a stainless steel bellows backed up by an external spring. Pressure applied within the bellows causes deflection of the spring and movement of a stem to which a differential transformer is mounted. This measures the pressure within the bellows and transmits an A. C. signal proportional to the pressure. The advantage of this pressure gauge is that the spring is outside of the coolant stream and thus operates at a fairly constant temperature.

A special type pressure gauge was also designed for use in the fuel rods. It consists of a small mu-metal electromagnet mounted above a mu-metal anvil actuated by a small copper bellows soldered into the top of the fuel rod. As the gas pressure in the rod changes, the bellows moves the anvil causing a change in the inductance in the magnet coil with a resultant change in resistance. The magnet is connected in a bridge circuit thus giving a change in voltage with a change in pressure in the rod.

In the metallurgical development of the fuel elements, Argonne was the first to use thermal cycling as a method for simulating reactor irradiation conditions. The technology of metallurgical

processing of uranium metal was also furthered so that the techniques and methods for casting, fabricating and heat treating which were developed for the EBR are now finding applications in production and other type reactors. The EBR is the only reactor in which fuel rods are finished to proper diameter without requiring additional machining or grinding, thus resulting in process savings.

The .448" O.D. tubes that contain the fuel rods in the active core have two ribs projecting longitudinally along the rod to assist in accurate spacing of the fuel (See Figure 3.) Difficulty was experienced in the first attempts to fabricate these rods to this shape. A hydraulic forming technique was devised whereby a tube of circular cross section is placed in a die containing longitudinal grooves. A rubber rod (the "hydraulic fluid") is then inserted in the tube. Pressure applied to the rubber rod forces the stainless steel tube into the grooves in the die to obtain the desired cross section.

An electromagnetic flow meter was developed to measure the coolant flow rate. This idea stemmed from a similar type flow meter that was being used by medical scientists for measuring the flow rate of blood in the human body.

These engineering developments are only a few of the many that were required before successful operation of the Experimental Breeder Reactor was achieved.

Operation

The machine has been operating regularly without serious difficulty. After the reactor had been operated for 374,000 kilowatt hours, the average burnup by fission in the core was found to be .0364% by measuring the cesium activity in sample fuel elements. Burnup in the central fuel element was .0432% and in an element at the outer edge of the core it was found to be .026%

Shortly after operation was begun, it was found that the shielding was inadequate and it was necessary to add 30" of concrete. The probable reasons for this underestimation of the shield thickness are discussed in the shielding section of this report. In June, 1952, a leak was detected between the primary and secondary coolant system and the reactor was shut down for repairs on June 6. Since the reactor was shut down for repairing the NaK leak and a sixty-day cooling period is required for the fuel, it was decided to remove 16 fuel elements selected from radial sections for processing even though only 0.0346% average burnup had been attained in the core. By taking advantage of the cooling of the fuel during the maintenance shutdown, it was possible to begin the analysis at an earlier date, without affecting the accuracy of the radio-chemical analysis. Additional U^{235} has been fabricated into rod form and loaded into the core to replace the samples removed. The NaK leak in the

heat exchanger has been repaired and the reactor is again in operation.

To date the reactor has produced over 1,500,000 kilowatt-hours of heat energy and supplies electricity for the building and associated facilities daily. The efficiency in producing electrical power has been 17½%.

Physics

The required critical mass for U^{235} is approximately 30% greater than previously estimated. Critical experiments performed at Los Alamos using basic geometrical configurations approximating that of the EBR indicated a value of approximately 40 kgs of U^{235}. Multigroup transport and diffusion theory, correcting for the difference between the Los Alamos experiments and the EBR, indicated a critical mass of 38-40 kg for the EBR; Monte Carlo calculations indicated a super-critical condition assuming 40 kg critical mass. However, the actual critical mass of the cold, clean core was found to be 51.8 kg.

The discrepancy between the actual and calculated critical mass for the U^{235} loading has not been completely resolved. It may in part be due to lack of accurate nuclear data such as cross sections in the EBR neutron spectrum, and the streaming of neutrons through coolant channels.

Control of the reactor is achieved by adjusting: (1) the vertical position of the external blanket, (2) twelve natural uranium control rods in the external blanket, and (3) a movable block of natural uranium in the outer blanket at the bottom of the reactor. Originally, there were also two boron rods in the inner blanket. These, however, were found to control such a small amount of reactivity that they were later removed and replaced by regular natural uranium blanket rods. The following table indicates the effectiveness of the various control methods and compares the calculated with the actual figures:

Method of Control	Calculated Worth Inhours	Actual Worth Inhours
Outer blanket	3080	3200
Top 4" of outer blanket	Not calculated	350
12 external control & safety rods	1000	120
Natural uranium block at bottom	448	27
Two boron rods	200	30

It is expected that the contribution of fissions in U^{238} to the breeding ratio will be considerably larger than was anticipated because the percentage of high energy flux in the tamper is greater than expected. It is estimated that 14% of the fissions occur in the U^{238} and the resulting contribution of the fast effect to the breeding gain is about 0.28. As a result of the greater number of fissions in U^{238} in the blanket than originally expected the power level is limited by the cooling capacity of the outer blanket.

Preliminary measurements of the flux distribution in the reactor have been made and indicate a flux of about 0.936×10^{14} neutrons/cm^2 in the center of the core, about 4.86×10^{13} at the edge of the core, 1.9×10^{12} between the internal and external blanket, and 1.2×10^{11} at the outer edge of the external blanket. The leakage has been found to be about 4%. The power distribution is approximately as follows:

Core	71.6%
Inner Blanket	14.3
Outer Blanket	14.1
	100.0%

The operating characteristics of the reactor appear to be stable with an overall negative temperature coefficient of -1.37 ih/°C.

Additional measurements of nuclear constants such as α and the generation of poisons and diluents are being investigated. Preliminary measurements indicate a value of 0.172 for α for U^{235} at the outer edge of the core and .123 in a sample 5.02 cm from the center of the core. These values are about twice the values originally estimated.

The following table shows the burnup and values of α found in some of the samples taken from the reactor after 374,000 kilowatt hours of operation.

Dist. from Center of core - cm.	% Burnup by Fission	% Burnup by Capture	α
0 (center rod)	.0432	-	-
2.51	.0414	-	-
5.02	.0374	.0047	.123
7.53	.0332	-	-
9.47	.0260	.0045	.172

The weighted average burnup after 374,000 kilowatt hours of operation was .0346%. The amount of burnup was determined by chemical analysis and measurement of the cesium activity in the sample fuel rods.

Two methods of determining the conversion ratio are being used, but neither analysis is complete as yet. It appears at this time, however, that the conversion ratio will be about 1.0.

Metallurgy

Slugs were .364" diameter and 1.875" long. Four such slugs are sandwiched between natural uranium in a stainless steel tube 22 mils thick with an outside diameter of .448". The assembled rods then contain 7½" of U^{235} in the center to make up the core, and 8" and 4-3/4" of natural uranium on the top and bottom respectively to make up the top and bottom sections of the inner blanket. The total length of fuel and fertile material then is 20¼".

A separate can 5 mils thick is used to contain the natural uranium to prevent dilution of the U^{238} into the U^{235}, while the U^{235} is inserted in the main tube without any other canning material around it. A NaK bond 0.020" thick is provided between the uranium rods and the outer stainless steel tube to provide a heat transfer bond.

The tubes lying outside the hexagonal metal separator in the inner blanket region are of natural uranium but could be used at any depletion. They are somewhat larger in diameter, .9375" O.D., and contain natural uranium jacketed in a 22 mil can. The length of U^{238} in the tubes is $20\frac{1}{4}$" while the total length of the tubes is $113\frac{1}{2}$" since they extend upward through the shield and are attached to the top plate.

Examination of the rods indicates that there has been no attack on the uranium and the wetting of the NaK bond to the fuel was not affected.

Plutonium-uranium alloy metallurgy is being initiated and it is anticipated that at some future date the EBR may be loaded with plutonium-uranium alloy and operate as a true breeder.

Shielding

The original shielding for the EBR was underestimated. The reasons for the difference between designed and actual performance is probably due to: (1) the increased 28 fissions in the blanket; (2) the small size of the reactor core which permits the exceed-

ingly high energy neutrons to escape; (3) streaming of neutrons in fuel channels; and (4) neutrons are of higher average energy than originally calculated, therefore more shielding is required to slow them down and capture them after they have left the reactor.

Only slight modification to placement of equipment outside the shield was required to provide space for the additional shielding.

Safety

Safe operation of the reactor is assured by numerous safety devices and interlocks to signal the operators in the event of improper performance of any of the equipment or to scram the reactor when necessary to prevent a possible runaway. There are at least 49 such safety devices. Twenty-six of these provide a signal to indicate abnormal operation, eight provide a danger signal and scram the reactor in two minutes if the trouble is not eliminated and 15 scram the reactor immediately. When trouble is detected the operator is immediately notified by a horn signal, and a light on the control board indicates the difficulty.

An immediate scram will take place in the event of any of the following parameters deviating from a predetermined value:

1. Positive pile period
2. Negative pile period

3. Reactor coolant flow rate
4. Reactor coolant inlet temperature
5. Reactor coolant outlet temperature
6. Heat exchanger outlet temperature
7. Fuel temperature
8. Elevator hydraulic pressure

9-10-11. Radiation level in three different positions.

Also, an immediate scram will occur if the following valves are not in the correct position:

1. Gravity tank outlet shut-off valve
2. Gravity tank outlet drain valve
3. Reactor overflow valve
4. Receiving tank overflow valve.

In addition to the automatic safety signals and scramming devices, extreme precautions are taken by the operators during the operation of the reactor. Frequent detailed inspections are made of all the equipment and safety devices and reported to the project engineer. Form A, attached to this report, is a check list used by the operators in periodic inspections and gives an indication of the precautionary measures taken.

Glossary

Blanket:	That part of the reactor surrounding the core and containing the fertile material.
Breeding gain:	The number of excess fissionable atoms produced per atom burned.
Breeding ratio:	The total number of fissionable atoms produced per atom burned.
Breeder reactor:	A reactor that produces the same kind of fissionable material that it burns, and further, produces more fissionable material than it burns.
Converter reactor:	A reactor that produces a different type of material than it burns.
Core:	That part of the reactor containing the active fuel.
Critical mass:	That mass of fuel required in the reactor to maintain a self-sustaining chain reaction.

Cross section: An expression of the probability of a certain nuclear reaction.

Doubling time: The time required to double the original fuel investment in a breeder.

Fast reactor: A reactor that does not contain a moderator to slow the neutrons down, but instead operates with high energy neutrons.

Fertile material: A material that, upon the absorption of a neutron, may be transmuted to a fissionable material. The fertile materials are Th^{232} and U^{238}.

Fissile material: Same as fissionable material.

Fission: The process that takes place when a neutron causes a fissionable atom to split in two or more fragments, giving off neutrons and gamma rays, and energy that is dissipated in the form of heat.

Fissionable material: A material that is capable of being fissioned. The fissionable materials include U^{233}, U^{235}, and Pu^{239}.

Fuel:	Same as fissionable material, i.e., a material that is capable of being fissioned.
Inhour:	That amount of reactivity required to make the pile period equal to one hour.
Intermediate energies:	The entire energy spectrum between thermal and fast.
Moderator:	A material that degrades the energy of a neutron when a collision occurs between the neutron and the moderator.
Pile period:	The time required to change the reactivity of the reactor by a factor of e.
Power density:	A measure of the power per unit of reactor core volume.
Reactivity:	A measure of the neutron multiplication.
Reflector:	A material surrounding the reactor for the purpose of reflecting neutrons back into the core.

Resonance peaks:	Points in the energy spectrum where the cross sections are higher than the general trend.
Specific power:	A measure of the power per unit of fuel mass.
Temperature coefficient:	A measurement of the change in reactivity relative to a unit change in temperature.
Thermal reactor:	A reactor that contains a moderator to slow the neutrons down to thermal energy, where the principal fission reactions occur.

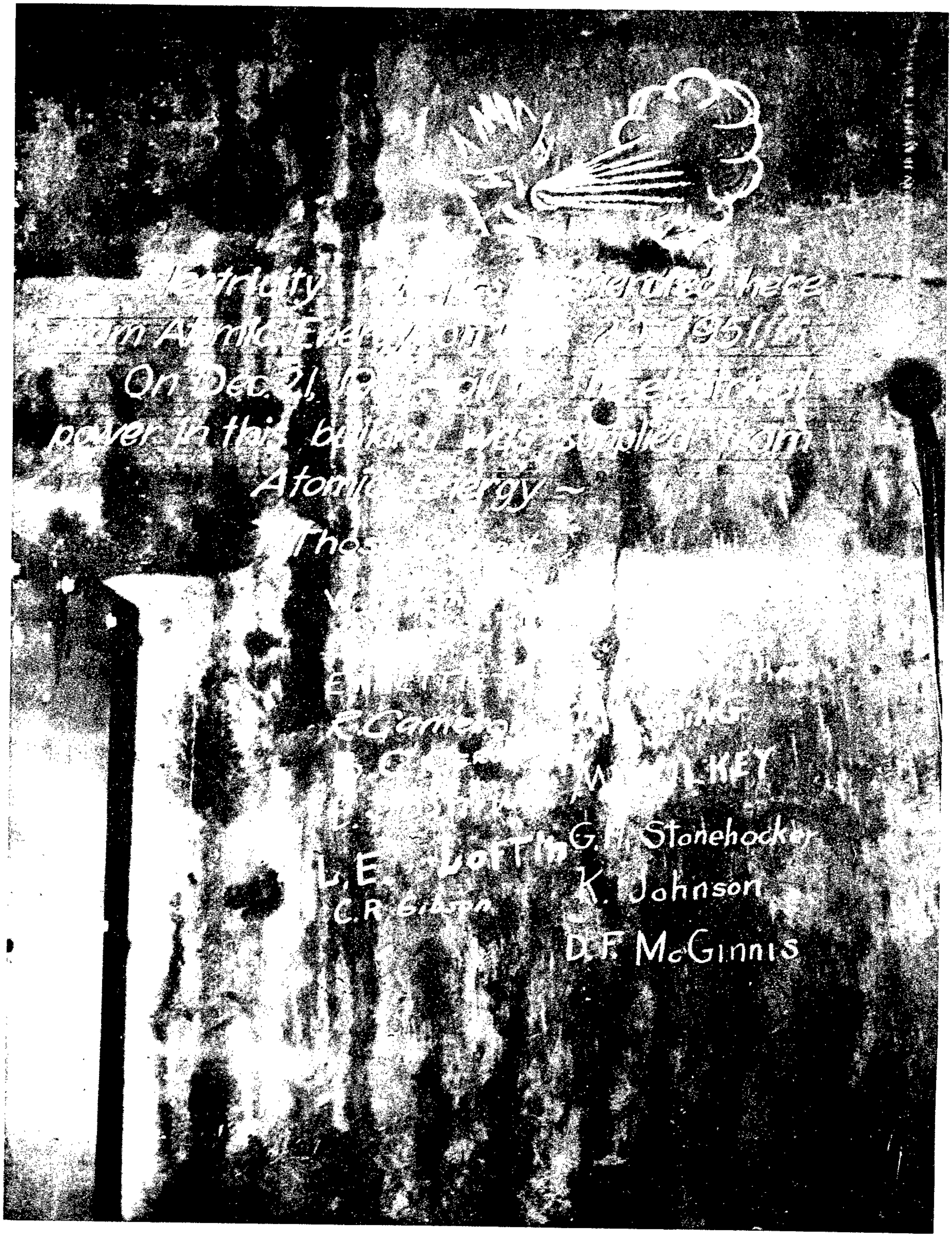

FIG. 1

FIG. 2

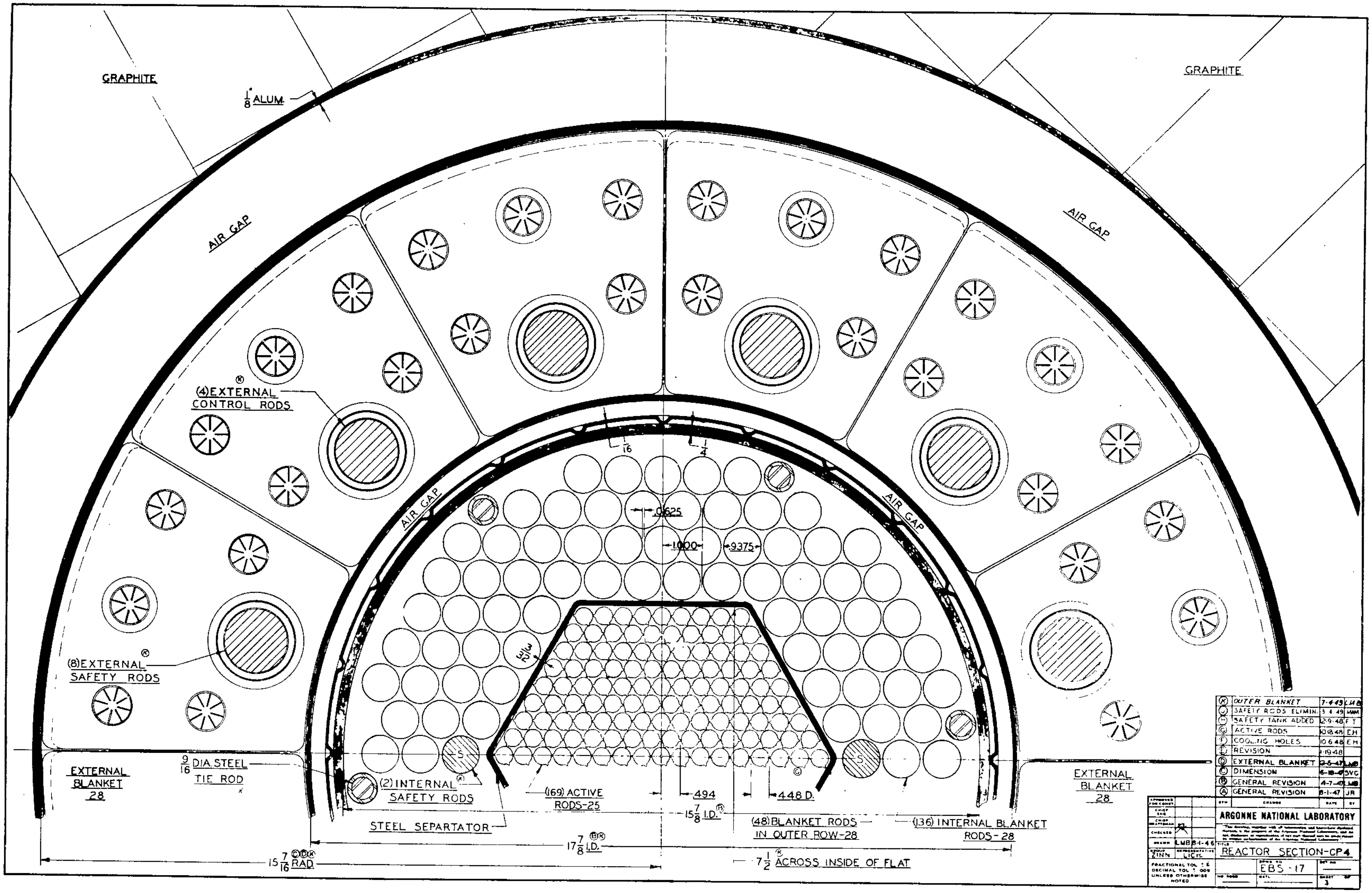
GRAPHITE
1/8" ALUM.
AIR GAP
GRAPHITE
AIR GAP
(4) EXTERNAL CONTROL RODS
(8) EXTERNAL SAFETY RODS
AIR GAP
AIR GAP
.0625
1.000
.9375
3/32
EXTERNAL BLANKET 28
9/16 DIA. STEEL TIE ROD
(2) INTERNAL SAFETY RODS
(169) ACTIVE RODS-25
.494
.448 D.
15 7/8 I.D.
(48) BLANKET RODS IN OUTER ROW-28
(136) INTERNAL BLANKET RODS-28
STEEL SEPARTATOR
17 7/8 I.D.
15 7/16 RAD.
7 1/2 ACROSS INSIDE OF FLAT
EXTERNAL BLANKET 28
OUTER BLANKET
SAFETY RODS ELIMIN.
SAFETY TANK ADDED
ACTIVE RODS
COOLING HOLES
REVISION
EXTERNAL BLANKET
DIMENSION
GENERAL REVISION
GENERAL REVISION
ARGONNE NATIONAL LABORATORY
REACTOR SECTION-CP4
EBS-17

FIG. 3

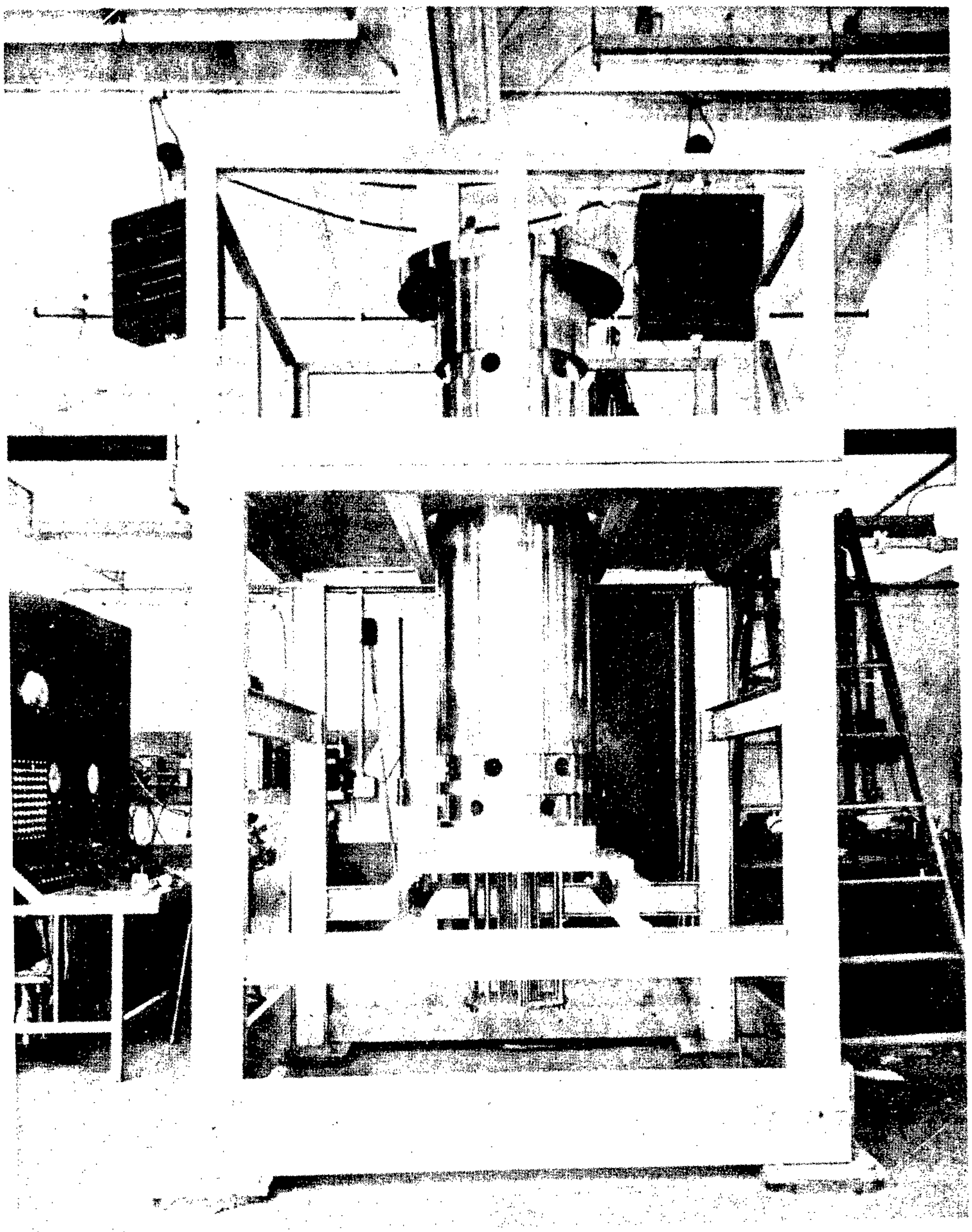

FIG. 4 Reactor Inner Assembly

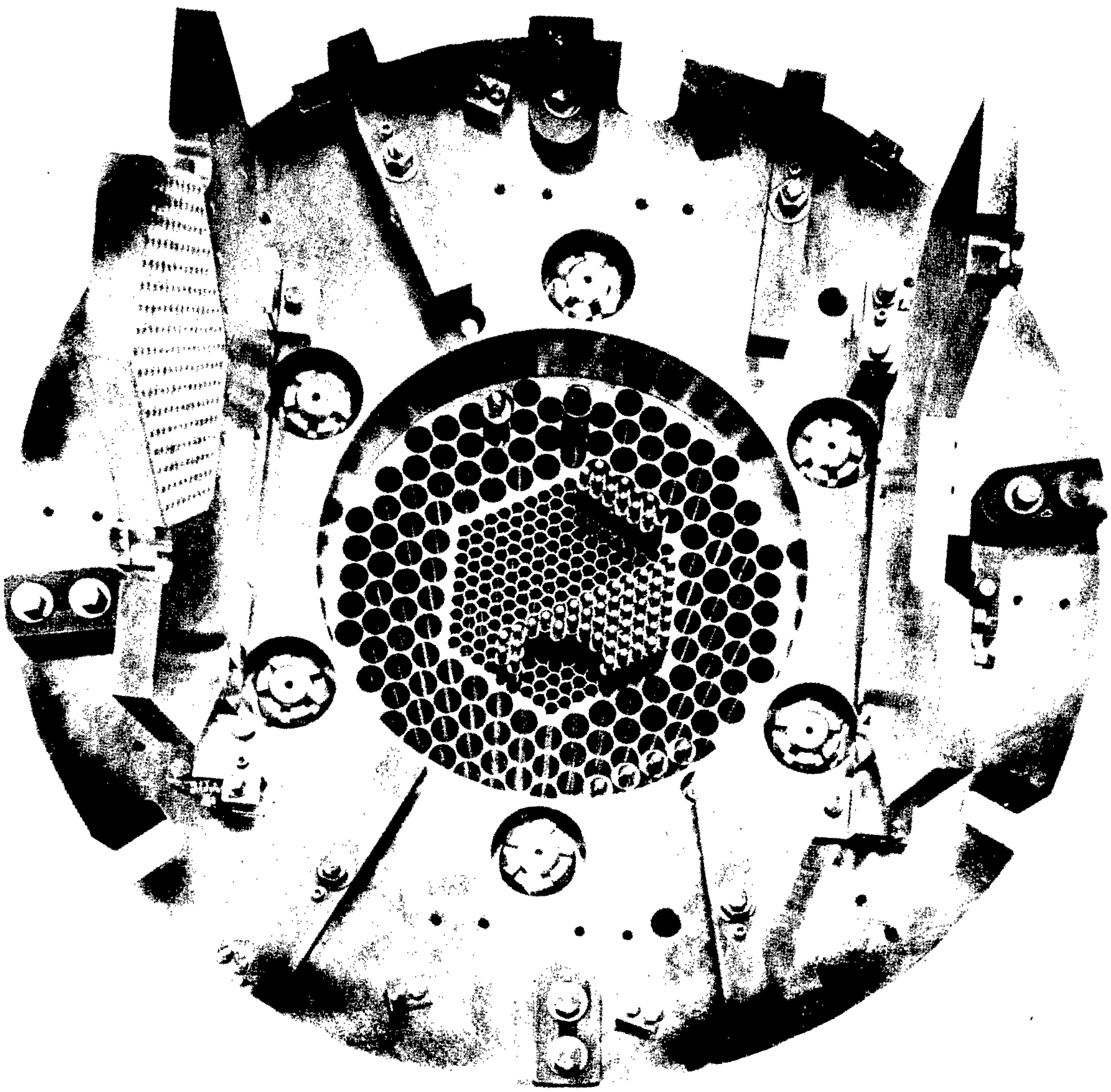

FIG. 5 Top of Assembly with Internal Doors Open

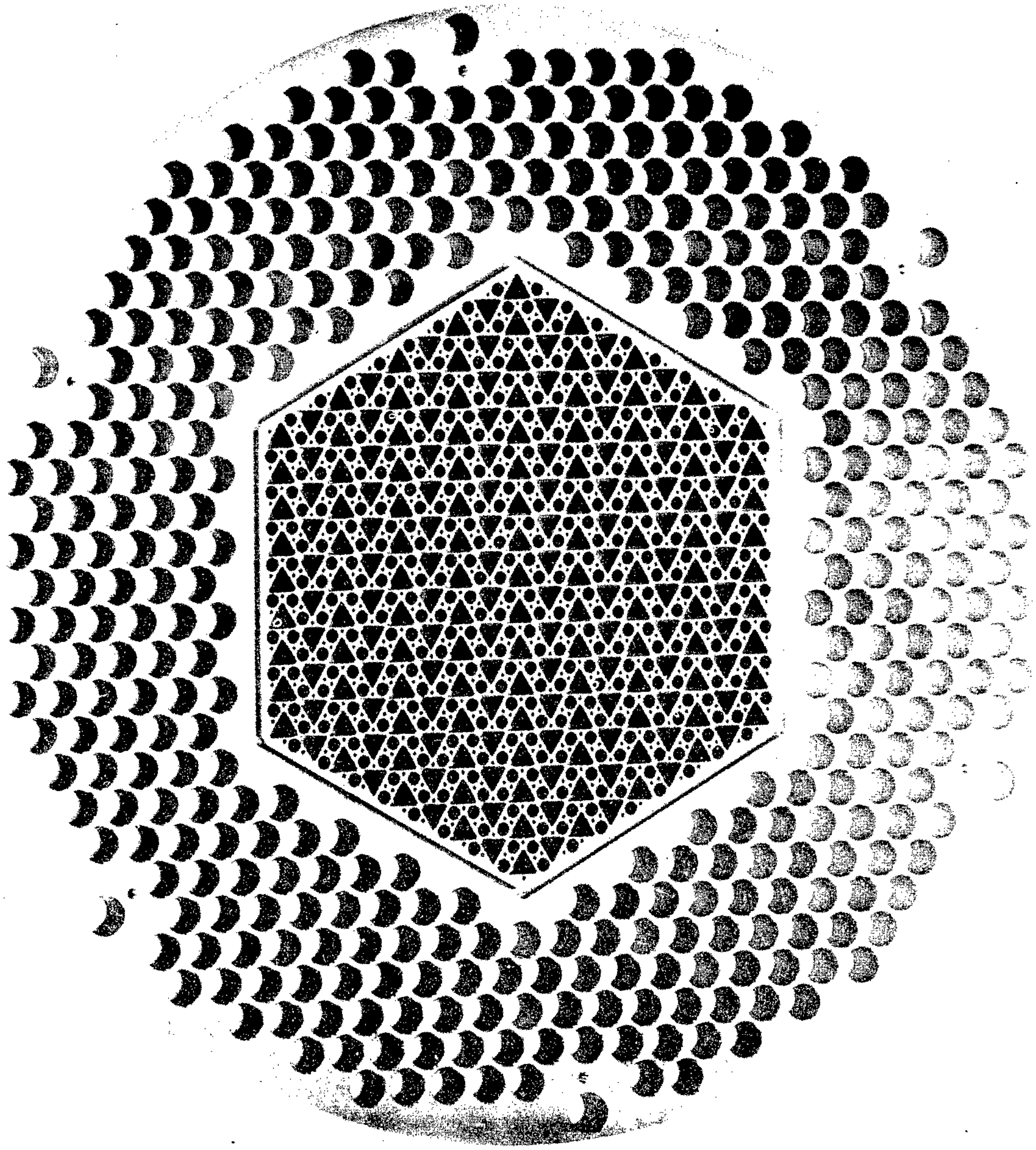

FIG. 6 Bottom Plate Showing Coolant Flow Holes and Rod Engaging Holes

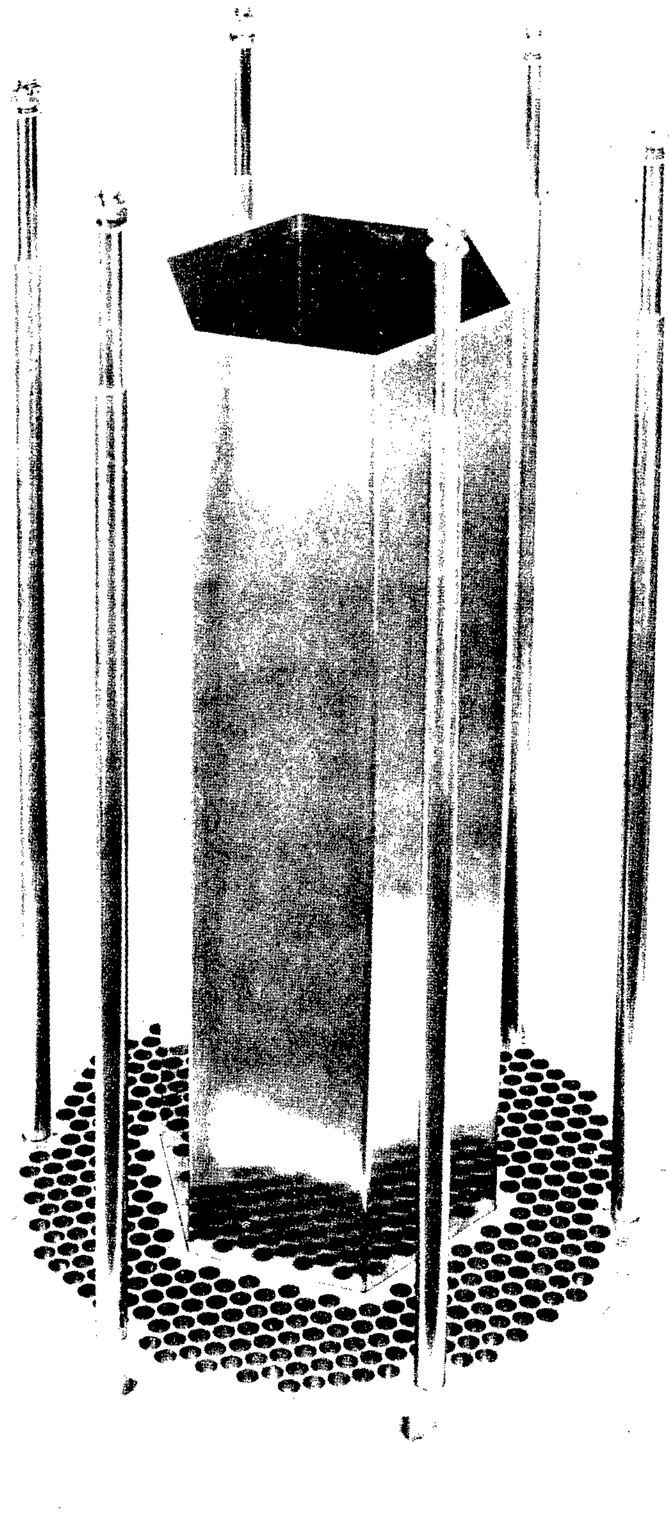

FIG. 7 Bottom Plate with Tie Rods and Coolant Flow Separator

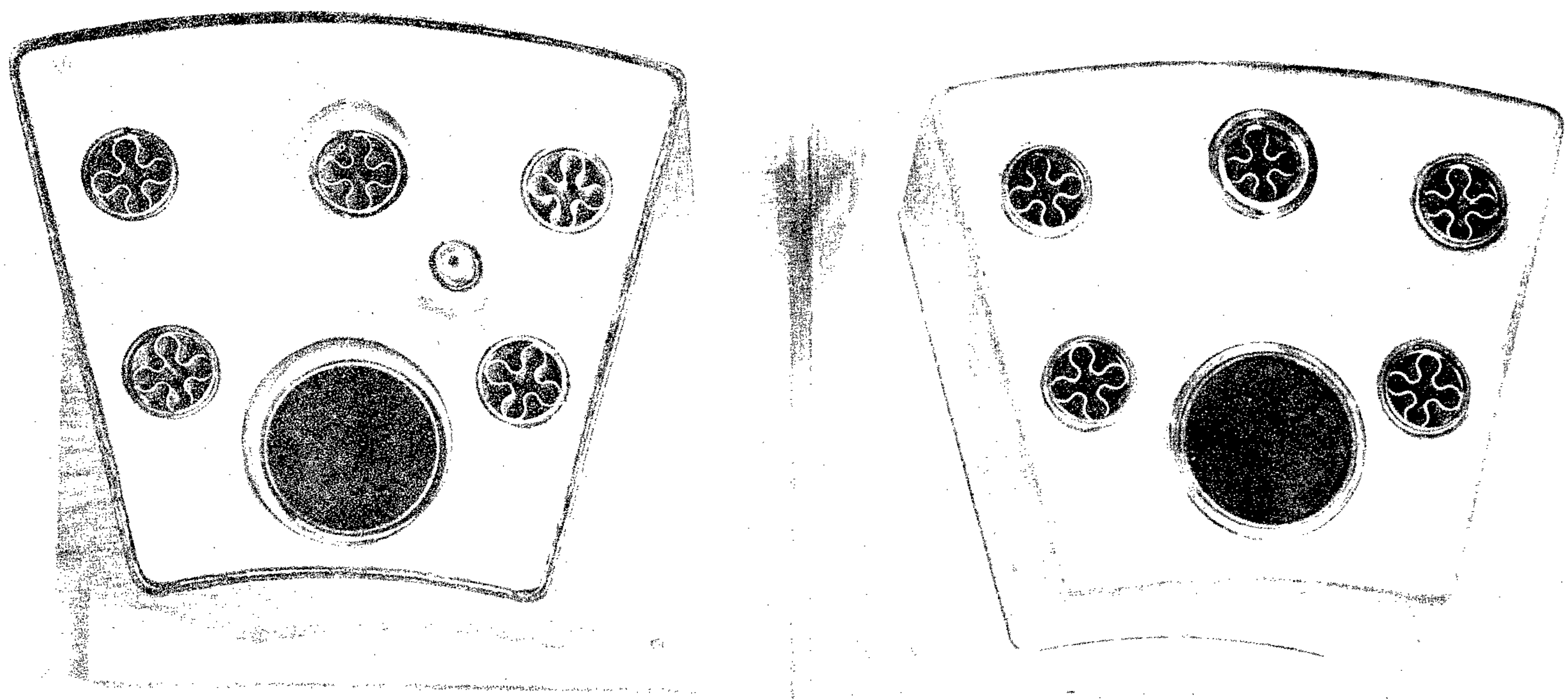

FIG. 8 Top and Bottom View of Blanket Brick

FIG. 9 Cup with Inner Liner in Place -- Side View

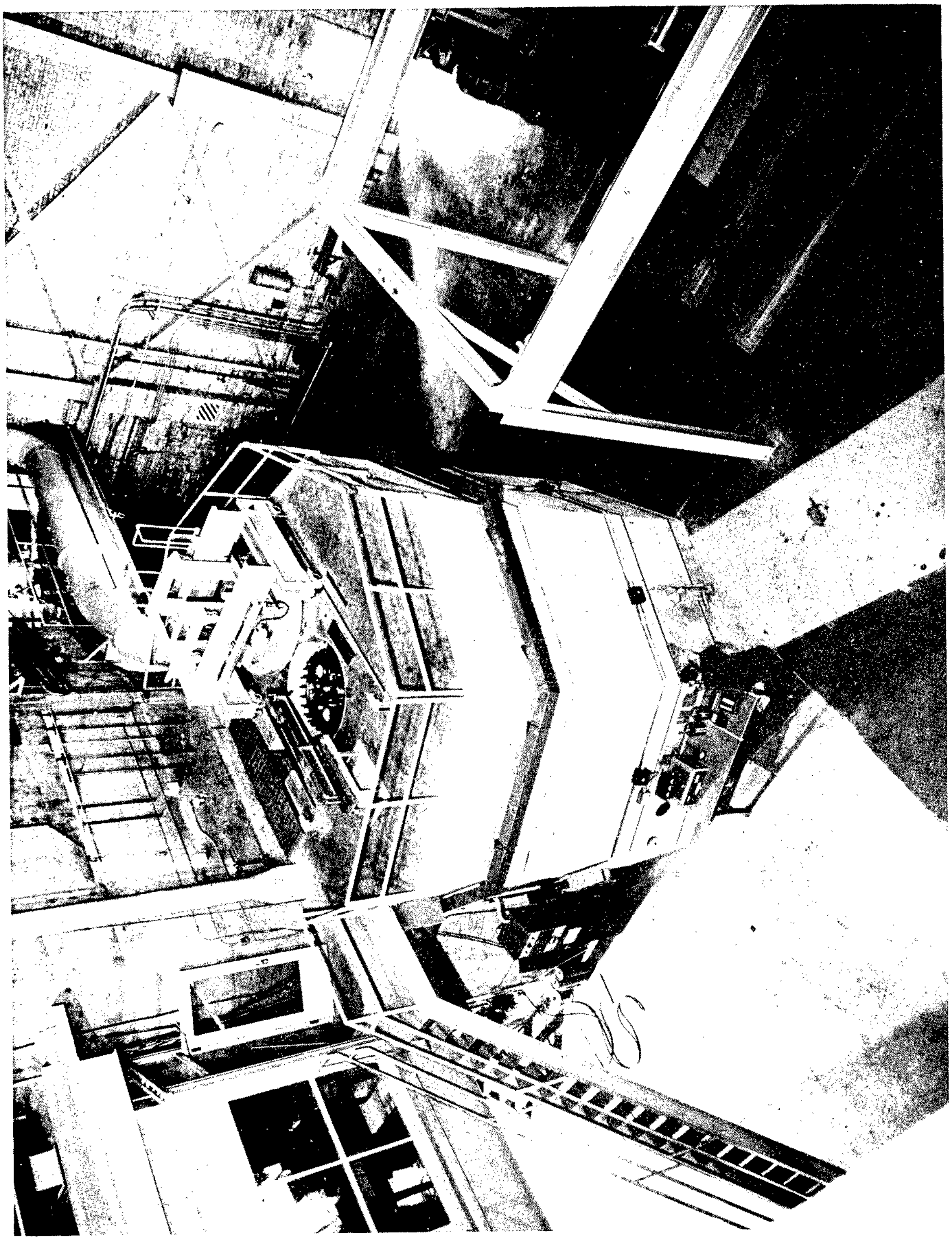

FIG. 10

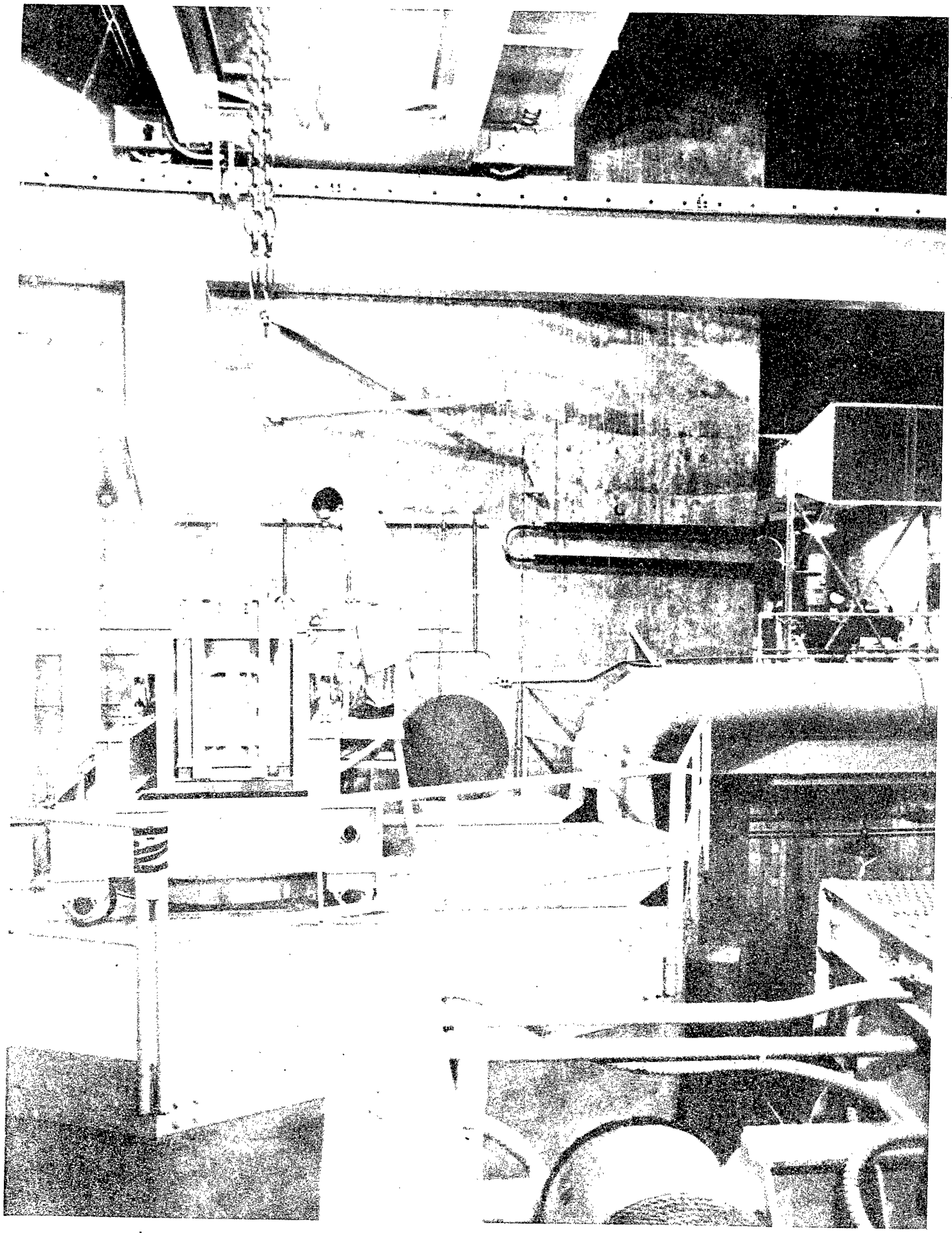

FIG. 11

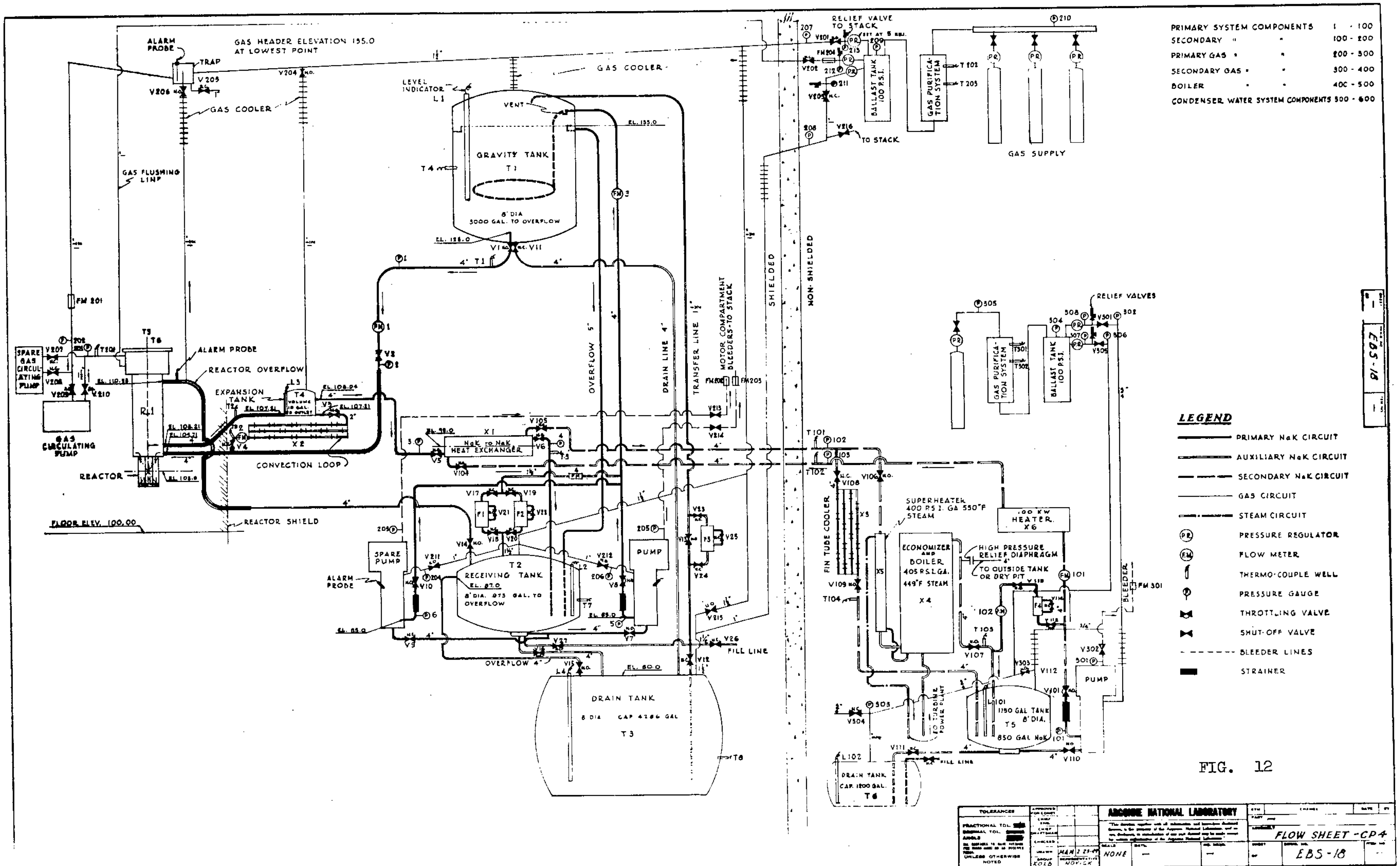
PRIMARY SYSTEM COMPONENTS 1 - 100
SECONDARY " " 100 - 200
PRIMARY GAS " " 200 - 300
SECONDARY GAS " " 300 - 400
BOILER " " 400 - 500
CONDENSER WATER SYSTEM COMPONENTS 500 - 600
LEGEND
PRIMARY NaK CIRCUIT
AUXILIARY NaK CIRCUIT
SECONDARY NaK CIRCUIT
GAS CIRCUIT
STEAM CIRCUIT
PRESSURE REGULATOR
FLOW METER
THERMO-COUPLE WELL
PRESSURE GAUGE
THROTTLING VALVE
SHUT-OFF VALVE
BLEEDER LINES
STRAINER
GAS HEADER ELEVATION 155.0 AT LOWEST POINT
ALARM PROBE
TRAP
GAS COOLER
GAS FLUSHING LINE
LEVEL INDICATOR
VENT
GRAVITY TANK
T1
8' DIA
3000 GAL. TO OVERFLOW
SPARE GAS CIRCULATING PUMP
GAS CIRCULATING PUMP
REACTOR
REACTOR OVERFLOW
EXPANSION TANK
CONVECTION LOOP
REACTOR SHIELD
FLOOR ELEV. 100.00
NaK to NaK HEAT EXCHANGER
X1
X2
OVERFLOW 5"
DRAIN LINE 4"
TRANSFER LINE 1½"
MOTOR COMPARTMENT BLEEDERS-TO STACK
SHIELDED
NON-SHIELDED
SPARE PUMP
PUMP
RECEIVING TANK
T2
8' DIA. 875 GAL. TO OVERFLOW
DRAIN TANK
8' DIA CAP 4286 GAL
T3
FILL LINE
RELIEF VALVE TO STACK
BALLAST TANK 100 P.S.I.
GAS PURIFICATION SYSTEM
TO STACK
GAS SUPPLY
RELIEF VALVES
FIN TUBE COOLER
SUPERHEATER 400 P.S.I. GA 550°F STEAM
ECONOMIZER AND BOILER 405 P.S.I.GA. 449°F STEAM
X4
HIGH PRESSURE RELIEF DIAPHRAGM
TO OUTSIDE TANK OR DRY PIT
100 KW HEATER X6
TO TURBINE POWER PLANT
1150 GAL TANK 8' DIA.
T5
850 GAL NaK
DRAIN TANK CAP. 1200 GAL.
T6
BLEEDER
FM 301
ARGONNE NATIONAL LABORATORY
FLOW SHEET - CP4
EBS-18
NONE

FIG. 12

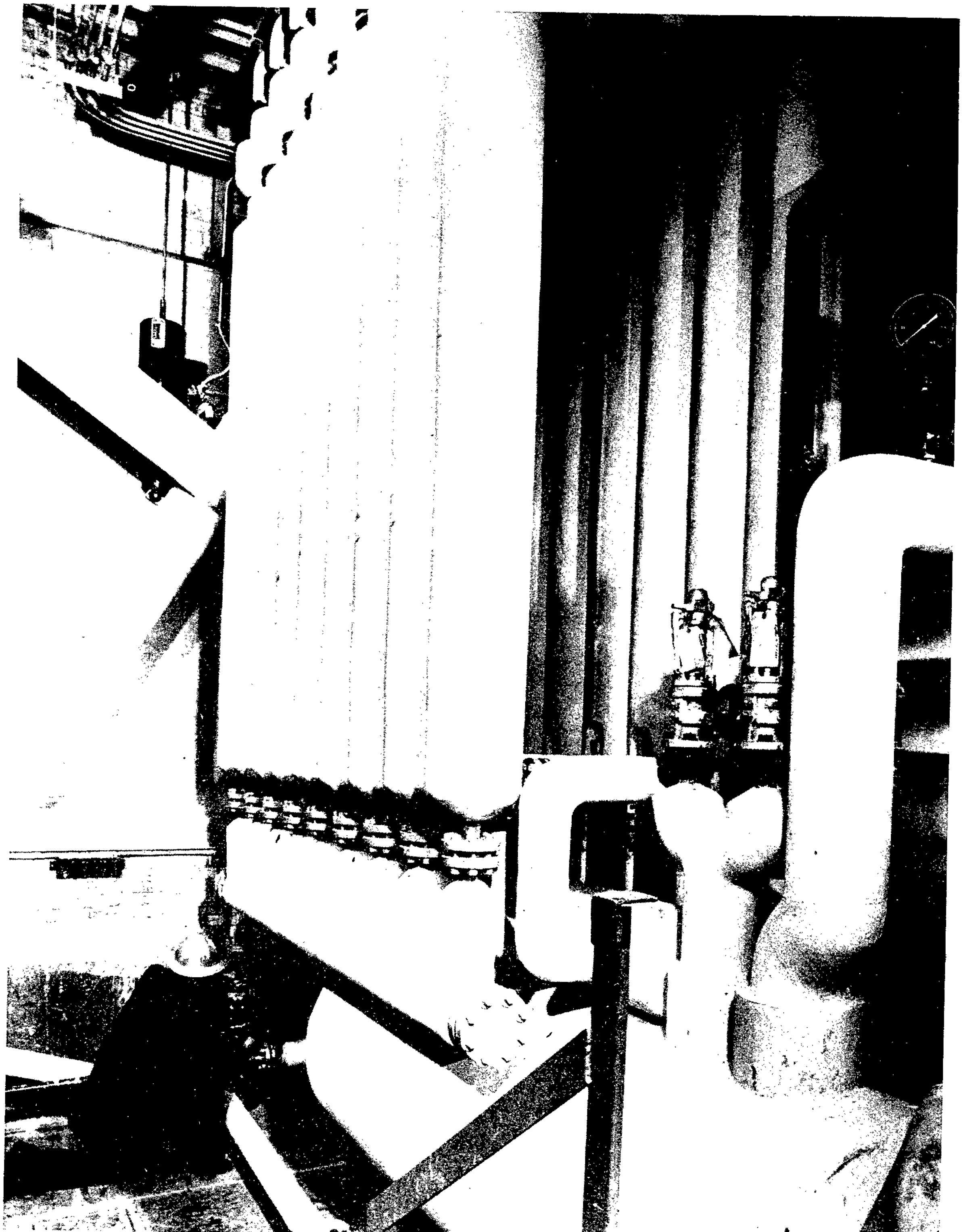

FIG. 13

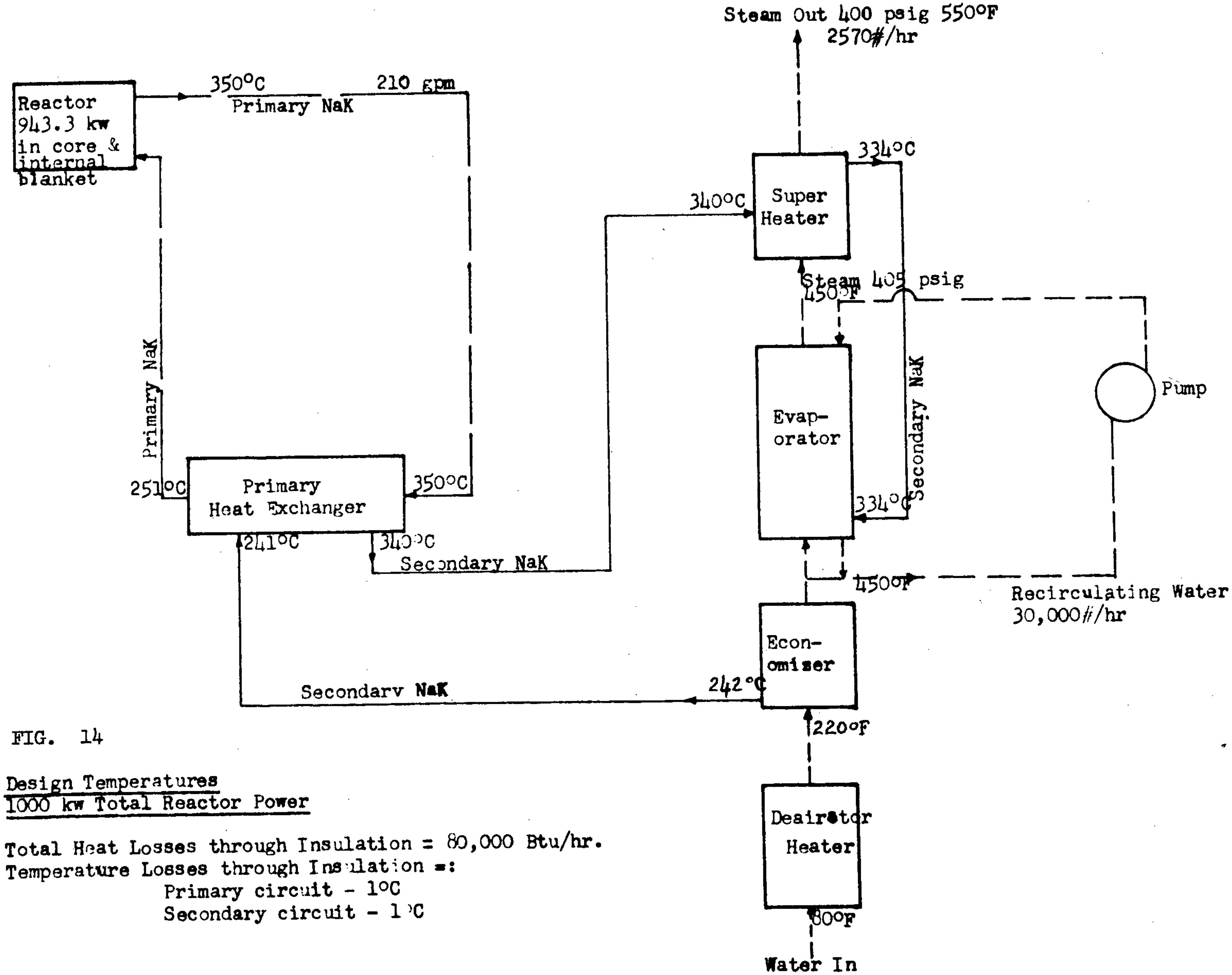

FIG. 14

Design Temperatures
1000 kw Total Reactor Power

Total Heat Losses through Insulation = 80,000 Btu/hr.
Temperature Losses through Insulation =:
Primary circuit - 1°C
Secondary circuit - 1°C

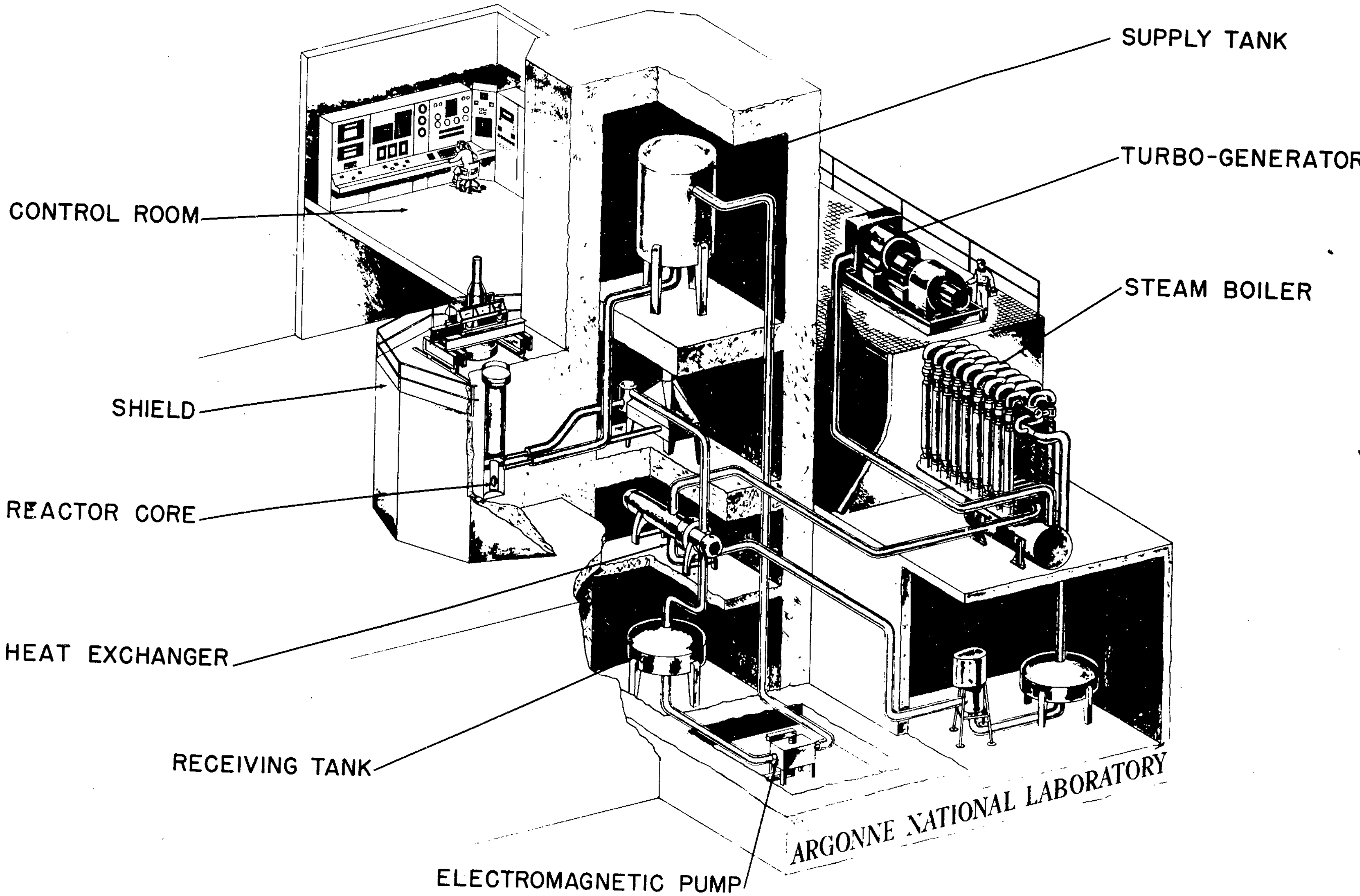
SUPPLY TANK
TURBO-GENERATOR
STEAM BOILER
CONTROL ROOM
SHIELD
REACTOR CORE
HEAT EXCHANGER
RECEIVING TANK
ELECTROMAGNETIC PUMP
ARGONNE NATIONAL LABORATORY

FIG. 15

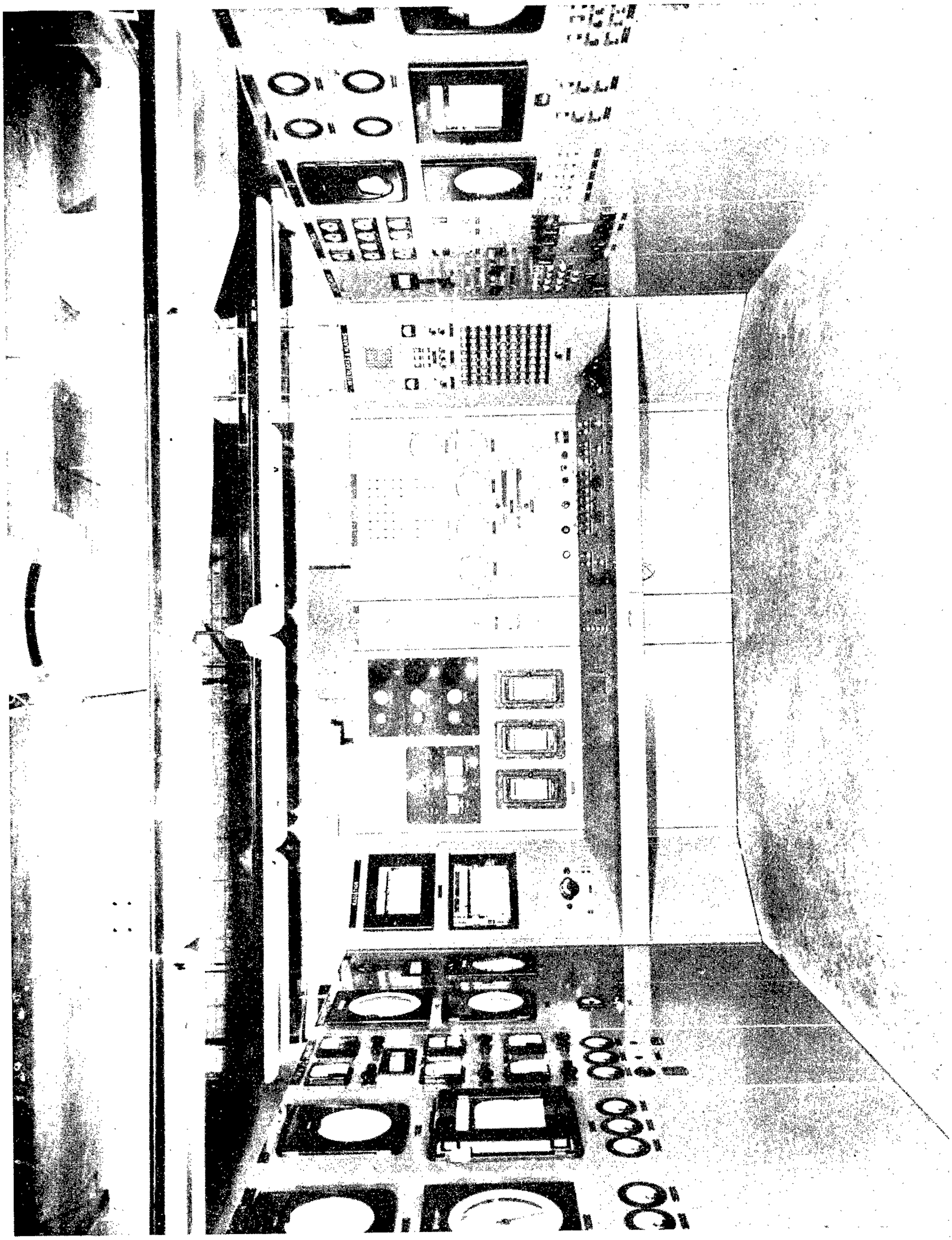

FIG. 16

Form A

Interlock Check on ____________, 1952

A. Reactor Section

1. Will safety rods go up with block down?____________________
2. Does tripping block trip safety rods?____________________
3. Will elevator go above 30" with safeties down?____________
4. Will tripping one safety rod stop elevator if above 30"?______
5. Will elevator go up with jacks above lower limit?____________
6. Will moving jacks off lower limit stop elevator if above 30"?___
7. Do jacks come down on "Reactor OFF" trip?__________________
8. Do jacks come down on "Reactor Shut Down" trip?____________
9. Setting on Safety Circuit #2 - tap number ______________
10. Setting on Safety Circuit #3 - tap number ______________
11. Trip level on #2__________KW. Previous trip level_________KW
12. Trip level on #2__________KW. Previous trip level_________KW
13. Trip level settings after testing #2 tap___, #3 tap__________
14. Period to trip Period Meter #1: On period meter___________sec.

    on Vibrating Reed______________sec.
15. Did #2 trip on negative period at scram?__________________
16. Period to trip Period #2: On period meter ______________sec.

    on Vibrating Reed______________sec.
17. Did #1 trip on negative period at scram?__________________
18. Previous trip levels #1______, ______; #2______, __________
19. Trip level settings after testing - #1_________, #2__________
20. Time to scram when automatic control goes above top power limit__

    _______min.
21. Time to scram when automatic control goes below lower power

    limit _______min.

22. Low hydraulic pressure trips interlock at ____________________psi.

23. Does reactor scram when elevator comes off top limit?____________

24. Low compressed air supply interlock trips at __________________psi.

25. Does cup probe interlock sound horn when lead is grounded?________

B. Coolant Section

1. Does pumped coolant flow alarm sound when flow falls below setting?

   ____

2. Normal setting of this interlock______________________________gpm

3. Does reactor coolant flow dropping below interlock setting scram reactor?____

4. Does reactor flow dropping below interlock setting open convection loop?____

5. Normal setting of this interlock______________________________gpm

6. Does power failure open convection loop?_______________________

7. Over-temperature of reactor inlet trips at ___________________°C.

8. Over-temperature of reactor outlet trips at __________________°C.

9. Over-temperature of heat exchanger outlet trips at ____________°C.

10. Over-temperature of fuel trips at _____________________________°C.

11. Reactor overflow trips at__________GPM at ___________________°C.

12. Previous overflow trip____________GPM at ___________________°C.

13. Does closing gravity tank outlet valve cause scram?____________

14. Does opening gravity tank drain valve cause scram?______________

15. Does closing reactor overflow valve cause scram?_______________

16. Does closing receiver tank overflow valve cause scram?___________

Form A

17. Does closing this valve also cause gravity tank shut-off valve to close?__________________________________

18. Does low level in gravity tank sound alarm?________________

19. Does secondary coolant flow dropping below interlock setting sound horn?__________________________________

20. Normal setting of this interlock____________________GPM.

21. Does mechanical primary pump under-speed sound horn?__________

22. Does secondary pump under-speed sound horn?________________

23. Do primary or secondary mechanical pump high temperatures sound horn?__________________________________

24. Does circulating gas flow falling below interlock setting cause delayed scram?______________________________

25. Low gas pressure in the primary system sounds horn at ________psi

26. Low gas pressure in the secondary system sounds horn at ______psi

27. Low gas pressure in the supply system sounds horn at ________psi

28. Do high levels in pumps sound horn?_____#1_________, secondary___

29. Does gas trap high level alarm sound horn?__________________

30. Does loss of boiler pump cooling water sound horn?____________

31. Does loss of cooling water on cup exhauster sound horn?_________

32. Does loss of cup cooling air cause delayed scram?_____________

33. Can reactor supply air be turned on before both graphite and cup exhaust are on?______________________________

34. Does shorting probe lead in primary EM pump sound horn?_________

35. Does shorting probe lead in transfer EM pump sound horn?________

36. Does smoke detector trip on smoke from cold pump room?_________

Form A

37. Does smoke detector trip sound horn?__________

38. Does liquid heater over-temperature sound horn?__________

C. General

1. List all manfunctions by letter and number above __________

   ____

2. Have these been corrected?__________

3. Have interlock panels been visually inspected for blocked relays, burned out coils, etc.?__________

4. Have all padlocks been replaced and locked?__________

This interlock check made by:

1. ____________________(shift supervisor)
2. ____________________
3. ____________________
4. ____________________

Approved by:

____

Project Engineer

☆ U. S. GOVERNMENT PRINTING OFFICE: 1956 O—377855